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SUBJECT: Application for amends to licenses DPR-58 & DPR-74,revising TS 3/4.6.2, "RCS Operational Leakage" by deleting Table 3.4-0, "RCS Pressure Isolation Valves," LCO 3.4.6.2f & Action c & SRs 4.4.6.2.2 for both units.

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AEP:NRC:1180

Donald C. Cook Nuclear Plant Units 1 and 2
Docket Nos. 50-315 and 50-316
License Nos. DPR-58 and DPR-74
TECHNICAL SPECIFICATION CHANGE REQUEST
REACTOR COOLANT SYSTEM PRESSURE ISOLATION VALVES

U. S. Nuclear Regulatory Commission
Document Control Desk
Washington, DC 20555

Attn: T. E. Murley

November 15, 1993

Dear Dr. Murley:

Reference:

1) USNRC Letter to all LWR Licensees: LWR Primary Coolant System Pressure Isolation Valves, February 25, 1980

2) AEP:NRC:0371, Reactor Coolant System Pressure Isolation Valves, March 24, 1980

3) Letter, S.A. Varga, USNRC to J. Dolan, Indiana and Michigan Electric Company, Order for Modification of Licenses Concerning Primary Coolant System Pressure Isolation Valves, April 20, 1981

4) WASH 1400/NUREG 75/014, Reactor Safety Study: An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants, USNRC, October 1975

5) AEP:NRC 1082E, Individual Plant Examination Submittal/Response to Generic Letter 88-20, May 1, 1992

6) AEP:NRC 1082F, Individual Plant Examination Response to NRC Questions, February 24, 1993

Adol
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This letter and its attachments constitute an application for a technical specification (T/S) change for Donald C. Cook Nuclear Plant Units 1 and 2. Specifically, we propose to change T/S 3/4.6.2, "Reactor Coolant System Operational Leakage," by deleting Table 3.4-0, "Reactor Coolant System Pressure Isolation Valves," Limiting Condition for Operation 3.4.6.2f and Action c., and Surveillance Requirements 4.4.6.2.2, for both units. We are also proposing three editorial changes in each unit's T/S, and the deletion of the last paragraph in the Bases for T/S 3/4.6.2.

The Reactor Coolant System pressure isolation valves in Table 3.4-0 of the Technical Specifications were added to the Limiting Condition for Operation on Reactor Coolant System Operational Leakage pursuant to a USNRC letter to all LWR licensees, "LWR Primary Coolant System Pressure Isolation Valves", dated February 25, 1980 (Reference 1) and following additional plant specific transmittals on this subject (References 2 and 3). Reference 1 was issued as a result of the finding in Reference 4 that an intersystem loss-of-coolant accident (ISLOCA) was a significant contributor to core damage frequency (Event V).

Recently, the Individual Plant Examination (IPE) (Reference 5) for Cook Nuclear Plant found that the contribution to overall core damage frequency from an ISLOCA was the least (0.08%) of all initiating event contributors. This was in large measure due to the design of potential Event V-sequence flow paths at Cook Nuclear Plant compared to that used in WASH-1400. The potential paths at Cook Nuclear Plant contain either three check valves, a combination of two check valves and a closed motor operated valve, or two closed motor operated valves, whereas the typical WASH-1400 flowpath contained only two check valves in series with a locked open valve as shown in Figure V-3 of Reference 4. The resultant IPE Event V-sequence initiating event frequency (described in References 5 and 6) was approximately an order of magnitude lower than the WASH-1400 value of 4.00 E-6 and was also about three orders of magnitude lower than the average accident initiating event frequency employed in the rest of the IPE. Additionally, the WASH-1400 analyses used the very conservative assumption that, when the in-series check valves fail, the pressurized low pressure piping also failed. A more realistic plant-specific scenario was modeled in the Cook Nuclear Plant IPE ISLOCA analyses, wherein existing plant design capabilities were used which reduced the effects of this accident (see response to question 3 in Reference 6). Thus, the Event V sequence was found to be an insignificant contributor, both to the Cook Nuclear Plant core damage frequency and off-site doses.

In addition to the above, this request is also being made because startup of the Cook Nuclear Plant Units 1 and 2 following refueling outages has been delayed on a number of occasions as a direct result of the unnecessarily restrictive testing requirements and acceptance criteria mandated by the Technical Specifications for the valves in Table 3.4-0. All of the subject valves will still be tested on a refueling outage frequency to the leakage limits of ASME XI under the IST program. This will continue to provide protection from the potential occurrence of an Event V accident type to ensure the health and safety of the public.

The proposed changes and our significant hazards consideration analysis are provided in Attachment 1. The proposed marked-up revised T/S pages are included in Attachment 2. Attachment 3 contains the typed proposed revised T/S pages.

We believe that the proposed changes will not result in (1) a significant change in the types of effluents or a significant increase in the amounts of any effluent that may be released offsite, or (2) a significant increase in individual or cumulative occupational radiation exposure.

These proposed changes have been reviewed by the Plant Nuclear Safety Review Committee and the Nuclear Safety and Design Review Committee.

In compliance with the requirements of 10 CFR 50.91(b)(1), copies of this letter and its attachments have been transmitted to the Michigan Public Service Commission and to the Michigan Department of Public Health.

This letter is submitted prusuant to 10 CFR 50.30(b) and, as such an oath statement is attahced.

Sincerely,

E E Fitzpatrick

E. E. Fitzpatrick
Vice President

eh

Attachments

STATE OF OHIO)
COUNTY OF FRANKLIN)

E. E. Fitzpatrick, being duly sworn, deposes and says that he is the Vice President of licensee Indiana Michigan Power Company, that he has read the forgoing Technical Specification Change Request Reactor Coolant System Pressure Isolation Valves and knows the contents thereof; and that said contents are true to the best of his knowledge and belief.

E E Fitzpatrick

Subscribed and sworn to before me this 15th

day of November, 19 93.

Diana L. Eads

NOTARY PUBLIC

DIANA L. EADS
Notary Public, State of Ohio
My commission expires 2-24-95

cc: A. A. Blind
G. Charnoff
J. B. Martin - Region III
NFEM Section Chief
NRC Resident Inspector - Bridgman
J. R. Padgett

bc: S. J. Brewer
J. B. Kingseed/J. M. Nieto/R. B. Bennett
D. H. Malin
J. D. Benes/E. V. Gilabert/J. J. Ripak
J. D. Grier/D. F. Powell/J. G. Nogrady
M. L. Horvath - Bridgman
J. B. Shinnock
W. G. Smith, Jr./S. H. Steinhart
B. A. Wetzel, NRC - Washington, DC
AEP:NRC:1180
DC-N-6015.1

STATE OF OHIO)
COUNTY OF FRANKLIN)

E. E. Fitzpatrick, being duly sworn, deposes and says that he is the Vice President of licensee Indiana Michigan Power Company, that he has read the forgoing Technical Specification Change Request Reactor Coolant System Pressure Isolation Valves and knows the contents thereof; and that said contents are true to the best of his knowledge and belief.

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Diana L. Eads

NOTARY PUBLIC
DIANA L. EADS
Notary Public, State of Ohio
My commission expires 2-24-95

ATTACHMENT 1 TO AEP:NRC:1180

DESCRIPTION OF PROPOSED CHANGES AND

10CFR50.92 SIGNIFICANT HAZARDS CONSIDERATION ANALYSIS

BACKGROUND

The Reactor Safety Study (WASH-1400/NUREG 75/014) (Reference 1) analyzed a so-called Event V Sequence resulting in an intersystem loss-of-coolant accident (ISLOCA). It was concluded that the ISLOCA was a significant contributor to core damage frequency since the containment is bypassed and reactor coolant is released directly to the Auxiliary Building.

As a result of the above finding, the USNRC issued a letter, "LWR Primary Coolant System Pressure Isolation Valves," dated February 25, 1980, (Reference 2) requesting LWR licensees to provide the following information:

1. Describe the valve configuration at your plant and indicate if an Event V isolation valve configuration exists within the Class I boundary of the high pressure piping connection PCS piping to low pressure system piping; e.g., (1) two check valves in series, or (2) two check valves in series with a MOV;

2. If either of the above Event V configurations exists at your facility, indicate whether continuous surveillance or periodic tests are being accomplished on such valves to ensure integrity. Also indicate whether valves have been known, or found, to lack integrity; and

3. If either of the above Event V configurations exist at your facility, indicate whether plant procedures should be revised or if plant modifications should be made to increase reliability.

AEPSC responded to the above USNRC letter with letter AEP:NRC:0371, "Reactor Coolant System Pressure Isolation Valves," dated March 24, 1980 (Reference 3). In this letter, the USNRC was informed that the following valve configurations are used at Donald C. Cook Nuclear Plant:

1) A minimum of three check valves in series

2) Two check valves with a minimum of a closed motor operated valve in series

3) Two check valves with a closed hand operated valve in series

4) A check valve with two closed air operated valves in series

AEP:NRC:0371 concluded that: "Therefore no Event V configuration exists at the Cook Plant. Consequently, the requests in Items 2 and 3 of Mr. Eisenhut's letter are not applicable to the Cook Nuclear Plant."

On April 20, 1981, the USNRC issued an "Order for Modification of Licenses Concerning Primary Coolant System Pressure Isolation Valves" for the Donald C. Cook Nuclear Plant (Reference 4). This order stated that "We have concluded that a WASH-1400 Event V valve configuration exists at your facility and that the corrective action as defined in the attached Order is necessary." Attached to the Order were the Technical Evaluation Report (TER) supporting the Order and the new Technical Specifications which, according to the Order, ". . .will ensure public health and safety over the operating life of your facility." These new Technical Specifications were incorporated at that time into the Operating Licenses for Cook Nuclear plant. These are the same Technical Specifications that we are now proposing to delete.

In 1987, the USNRC issued Generic Letter (GL) 87-06, "Periodic Verification of Leak Tight Integrity of Pressure Isolation Valves" (Reference 5). In our response to that GL, AEP:NRC:1041 (Reference 6), we stated that 12-SI-170L2, 12-SI-170L3, 12-RH-133, and 12-RH-134 are leak tested as per the surveillance requirements in the Technical Specifications. The proposed changes in this submittal, therefore, modify our response to GL 87-06.

Finally, in 1992, the USNRC issued Information Notice 92-36, "Intersystem LOCA Outside Containment" (Reference 7). This information notice listed eleven "Observed Plant Vulnerabilities to ISLOCA Precursors." A review of this information notice concluded that the concerns raised are adequately addressed by controls currently in place or planned at the Cook Nuclear Plant. The proposed changes to the Technical Specifications will not alter this conclusion because our response to the information notice was not based on existing technical specifications.

COOK NUCLEAR PLANT IPE

On May 1, 1992, AEPSC responded to Generic Letter 88-20 (Reference 8), in AEP:NRC:1082E, Individual Plant Examination (IPE) submittal (Reference 9) and provided further information on the analysis of the ISLOCA in Reference 10. (See the Enclosure to this Attachment: Donald C. Cook Nuclear Plant, Paths Considered as Possible Event V-Sequence LOCA, taken from Reference 10). In the determination of the ISLOCA initiating event frequency, leak testing of 12-SI-170L2, 12-SI-170L3, 12-RH-133, and 12-RH-134, as required by the surveillance requirements in the Technical Specifications, was addressed. The IPE for Cook Nuclear Plant concluded that the ISLOCA was the accident that contributed the least to the overall core damage frequency. The calculated ISLOCA initiating event frequency was approximately 6.70E-07, and the calculated probability of ISLOCA core damage was approximately 5.4E-08. The contribution to overall core damage frequency from an ISLOCA is less than 0.1% of the 6.26E-05 IPE calculated core damage frequency per reactor year. It is seen, therefore, that the contribution from an ISLOCA event at Cook Nuclear Plant is negligible.

A COMPARISON OF WASH-1400 VS. COOK NUCLEAR PLANT WITH RESPECT TO ISLOCA

The ISLOCA accident is described in Section 5.3.2.5 of WASH-1400, and the quantification of its core damage frequency is found in Appendix V, Section 4.4 of WASH-1400.

The core damage frequency calculated in WASH-1400 is 4.00-06/reactor year. WASH-1400 evaluated 3 pathways of 2 check valves in series and assumed that the 600 psi Low Pressure Injection System (LPIS), once exposed to RCS pressure, would fail and create approximately a 6" effective diameter LOCA. No other accident initiation features or mitigating actions were modeled.

In the Cook Nuclear Plant IPE, nine different pathways for an ISLOCA event were analyzed. These pathways (see Enclosure to this Attachment), consisted of either three check valves in series, two check valves and a closed motor operated valve (MOV) in series, or two normally closed MOVs in series. Although WASH-1400 looked at check valve leak testing as a sensitivity analysis, the Cook Nuclear Plant IPE accounted for the in-place leak testing when calculating the probability of an ISLOCA occurring. In addition, the Cook Nuclear Plant IPE modeled mitigating actions for the ISLOCA event in the event tree. The

IPE ISLOCA initiating event frequency was approximately an order of magnitude lower than the WASH-1400 core damage frequency (4.00E-06) and the IPE ISLOCA core damage frequency value was approximately two orders of magnitude lower than the WASH-1400 values.

Finally, to address the impact of removing the T/S requirements and only testing 12-SI-170L2, 12-SI-170L3, 12-RH-133, and 12-RH-134 at a refueling outage frequency, a sensitivity run was made in support of this submittal. It was found that the ISLOCA initiating event frequency, and ISLOCA probability of core damage would also increase by 5.4% to the mid 5.00E-08 range, and the overall core damage frequency would remain unchanged.

IST PROGRAM

The subject check valves of this submittal are currently being tested in Mode 5, Cold Shutdown, as required by the Technical Specifications surveillance requirements prior to going to Mode 4, Hot Shutdown. The allowable leak rate cannot exceed 1 gpm. If these check valves are removed from the Technical Specifications, they would continue to be tested under the IST Program (ASME Boiler and Pressure Vessel Code, Section XI), on a refueling outage frequency, like their sister valves in the other loops in the Residual Heat Removal System. The acceptance criteria for the leak testing of these valves under the IST Program is 5 gpm.

DESCRIPTION OF PROPOSED TECHNICAL SPECIFICATION CHANGES

The proposed changes are listed below. They are identical for both units.

1. T/S 3.4.6.2 d. Add "and" at the end of the line.

2. T/S 3.4.6.2 e. Delete "and," and add a period after "gpm[2]".

3. T/S 3.4.6.2 f. Delete in its entirety.

4. T/S 3.4.6.2 APPLICABILITY: Delete "**" and replace with a "*".

5. T/S 3.4.6.2 ACTION c. Delete in its entirety.

6. T/S 3.4.6.2 ACTION c, footnote *. Delete the footnote.

7. T/S 3.4.6.2 APPLICABILITY, footnote **. Change the "**" to "*".

8. T/S 4.4.6.2.2. Delete in its entirety.

9. T/S 3/4.6.2, Table 3.4-0. Delete the Table and the footnote (a) in its entirety.

10. T/S 3/4.6.2, Bases. Delete the last paragraph in its entirety.

10CFR50.92 SIGNIFICANT HAZARDS CONSIDERATION ANALYSIS

Per 10 CFR 50.92, a proposed amendment to an operating license will not involve a significant hazards consideration if the proposed amendment satisfies the following three criteria:

1) Does not involve a significant increase in the probability or consequences of an accident previously analyzed,

2) Does not create the possibility of a new or different kind of accident from an accident previously analyzed or evaluated, or

3) Does not involve a significant reduction in a margin of safety.

Criterion 1

The ISLOCA is not one of the accidents previously analyzed in Chapter 14, Safety Analysis, of the Cook Nuclear Plant Updated Final Safety Analysis Report. Chapter 14 analyzes the large break LOCA in Section 14.3.1, and "loss of reactor coolant from small ruptured pipes or from cracks in large pipes which actuates the ECCS", or small break LOCA in Section 14.3.2. Therefore, deleting from the Technical Specifications the Reactor Coolant System pressure isolation valves in Table 3.4-0, will not increase the probability or the consequences of the large break or the small break LOCAs previously analyzed for the Cook Nuclear Plant.

Criterion 2

The Reactor Coolant System pressure isolation valves in Table 3.4-0 of the Technical Specifications were added because WASH-1400 identified the ISLOCA as a significant contributor to core damage frequency. Deletion of the subject valves from the Technical Specifications and reliance on the testing requirements mandated by the In-Service Testing Program of ASME XI does not create the possibility of a new or different kind of accident from the large break or the small break LOCAs previously analyzed for the Cook Nuclear Plant.

Criterion 3

Deleting the Reactor Coolant System pressure isolation valves from the testing requirements in Table 3.4-0 of the Technical Specifications will result in these valves only being tested on a refueling outage frequency as part of the ASME B&PV Code Section XI IST Program. This somewhat reduced testing frequency will result in a slight increase in the ISLOCA initiating event frequency, and ISLOCA contribution to core damage frequency of 5.4%, from lower 5.00E-08/reactor year to mid 5.00E-08/reactor year. This insignificant increase will not affect the overall core damage frequency of 6.26E-05/reactor year. Therefore, it is concluded that the proposed deletion of the Reactor Coolant System pressure isolation valves in Table 3.4-0 of the Technical Specifications, as well as the proposed deletion of the portions of the Technical Specifications that are affected by Table 3.4-0, will not result in a significant reduction in the margin of safety that exists at Cook Nuclear Plant to prevent an ISLOCA or to mitigate the consequences of an ISLOCA.

REFERENCES

1. WASH-1400/NUREG 75/014, Reactor Safety Study: An Assessment of Accident Risks in U.S. Commercial Nuclear Plants, USNRC, October 1975.

2. USNRC Letter to all LWR Licensees: LWR Primary Coolant System Pressure Isolation Valves, February 25, 1980.

3. AEP:NRC:0371, Reactor Coolant System Pressure Isolation Valves, March 24, 1980.

4. Letter, S. A. Varga, USNRC to J. Dolan, Indiana and Michigan Electric Company, Order for Modification of Licenses Concerning Primary Coolant System Pressure Isolation Valves, April 20, 1981.

5. USNRC Generic Letter (GL) 87-06, " Periodic Verification of Leak Tight Integrity of Pressure Isolation Valves," March 13, 1987.

6. AEP:NRC:1041, Generic Letter 87-06, Periodic Verification of Leak Tight Integrity of Pressure Isolation Valves, November 12, 1987.

7. USNRC Information Notice 92-36, "Intersystem LOCA Outside Containment", May, 7, 1992.

8. Initiation of the Individual Plant Examination for Severe Accident Vulnerabilities - 10CFR50.54(f), Generic Letter No. 88-20, November 23, 1988.

9. AEP:NRC1082E, Individual Plant Examination Submittal, Response to Generic Letter 88-20, May 1, 1992.

10. AEP:NRC 1082F, Individual Plant Examination Response to NRC Questions, February 24, 1993.

ENCLOSURE

DONALD C. COOK NUCLEAR PLANT PATHS CONSIDERED AS POSSIBLE V-SEQUENCE LOCAS

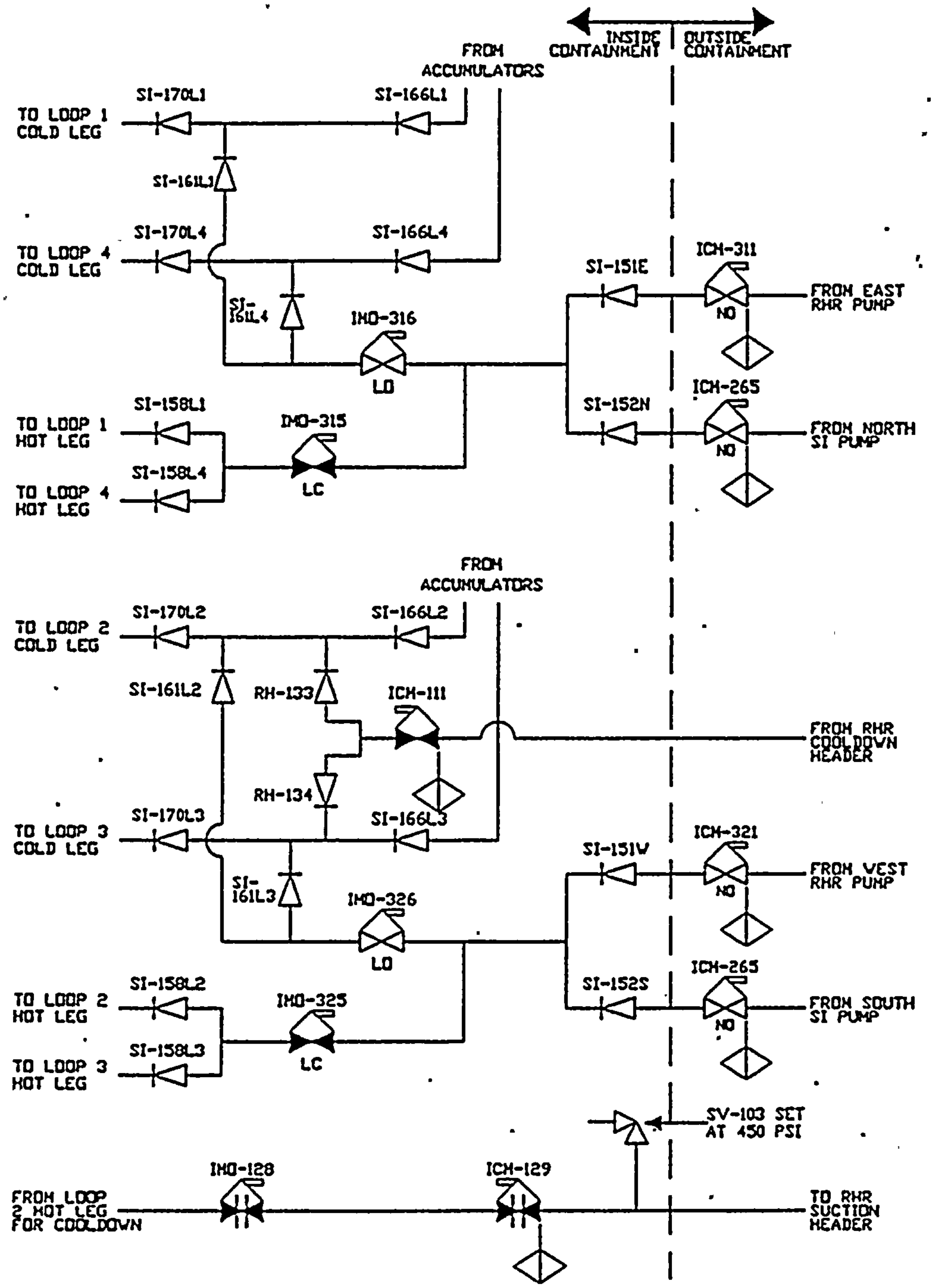

FIGURE 11

REFERENCE DRAWING: OP-1-5143-20